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Endorsement of ASME Code Section III Division 5: Filling Industry Needs and Gaps for Design and Construction of High Temperature Reactor Components

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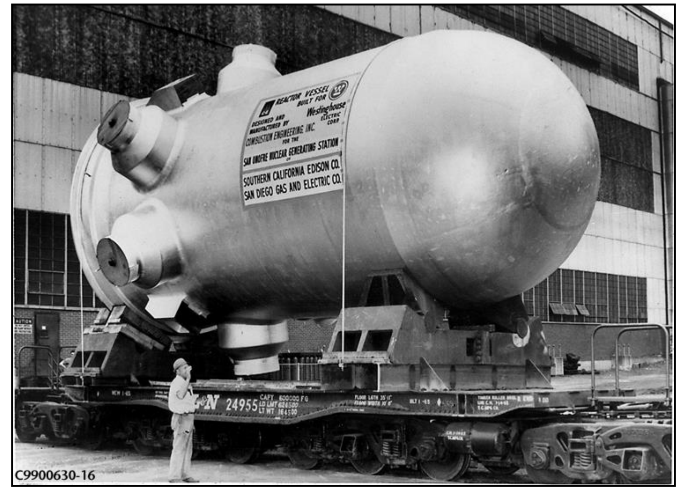
ASME Section III Treats Metallic Materials for Low & High Temperatures Differently

- **Allowable stresses for LWR & low-temperature advanced reactor components not time dependent**
  - **< 700°F (371°C) for ferritic steel and < 800°F (427°C) for austenitic matls**



PWR RPV

Monju SFR IHX

- **At higher temps, materials behave inelastically and allowable stresses are explicit functions of time & temp**
  - **Must consider time-dependent phenomena such as creep, creep-fatigue, relaxation, etc.**
  - **ASME Sec III Division 5 provides rules for construction of high temperature reactor components**

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ASME Section III Division 5, Specifically for High Temperature Reactors, Was First Issued in Nov 2011, revised in 2013 & 2015

- Sec III Div 5 contains construction and design rules for high-temperature reactors, including gas-, metal- & salt-cooled reactors
- Covers low temperature metallic components, largely by reference to other portions of Sec III
- Covers high-temperature metallic components explicitly, including former
  - Sec III, Subsections NG (Core Supports) & NH (Elevated Temperature Components)
  - Relevant Code Cases addressing time-dependent behavior and new materials
- Includes rules for graphite & ceramic composites for core supports & internals for first time in any international design code
- Numerous technical deficiencies in Div 5 have been identified. Some have been and others are being addressed

Construction Rules for Components of High Temperature Reactors Need to Be Updated and Endorsed

- NRC has begun to assess Sec III Div 5 for endorsement
  - Very important since predecessor ASME rules never endorsed
  - Will facilitate HTR design & applications and enhance regulatory surety for new licensees
- ASME is actively updating Div 5 and has formed task groups to support regulatory endorsement
- DOE's Advanced Reactor Technologies Office is conducting R&D supporting Div 5 endorsement issues including:
  - Resolution of numerous identified shortcomings in high temperature design methods (see review summaries)
  - Extension of materials allowables from 300,000 to 500,000 hrs to support 60-yr lifetimes of advanced reactors
  - Inclusion of graphite and & ceramic composites for core supports & internals for first time in any international design code

ART Materials Program Also Provides Technical Basis for ASME Division 5 Additions

- **Additional Materials are being added to ASME Division 5**
  - **Alloy 617, high-temperature nickel-based alloy to allow higher temperature heat exchangers and steam generators**
  - **Alloy 709, super stainless steel, to significantly improve high temperature strength and expand design envelop, performance, safety, and economics for advanced reactors**
  - **Hastelloy N (proposed) high nickel alloy compatible with salt-cooled reactors**
- **Additional high temperature design methods are being added to Division 5**
  - **Improved design rules at very high temperatures based on idealized elastic perfectly plastic (E-PP) material behavior**
  - **Rules for use of compact heat exchangers for improved power conversion efficiencies**
  - **Rules for high-temperature weld overlay clad structures (proposed) for use of currently qualified ASME Div 5 materials and compatibility with salt-cooled reactors**

ART Materials Program Also Provides Technical Basis for Additional Regulatory Requirements

- Corrosion studies of materials for usage with high temperature reactor coolants
  - Evaluation of alloys, graphite, and composites in HTGR helium chemistries and air/steam ingress
  - Evaluation of current Code-qualified and proposed new alloys for compatibility with sodium for fast reactor applications
  - Evaluation of current Code-qualified and advanced materials for compatibility with salt-cooled reactors
- Development and validation of irradiation-effects models
  - Qualification of irradiation and irradiation-creep effects for graphite behavior for ASME Section III Division 5
  - Development of validated models for predicting very high dose neutron exposures for fast reactor applications using ion irradiations

High Temperature Design Methods and Materials in ASME Div 5 Need Updating*

- **Weldments**
  - **Weldment evaluation methods, metallurgical & mechanical discontinuities, transition joints, tube sheets, validated design methodology**
- **Aging & environmental issues**
  - **Materials aging, irradiation & corrosion damage, short-time over-temperature/load effects**
- **Creep and fatigue**
  - **Creep-fatigue (C-F), negligible creep, racheting, thermal striping, buckling, elastic follow-up, constitutive models, simplified & overly conservative analysis methods**
- **Multi-axial loading**
  - **Multi-axial stresses, load combinations, plastic strain concentrations**

Based on Multiple DOE, NRC & National Lab Reviews of High Temperature Reactor Issues over Past 40 Years

High Temperature Design Methods and Materials in ASME Div 5 Need Updating *(cont)*

- **Materials allowables**
  - Elevated temperature data base & acceptance criteria, min. vs ave. props, effects of melt & fab processes, 60-year allowables
- **Failure criteria**
  - Flaw assessment and leak before break procedures
- **Analysis methods and criteria**
  - Strain & deformation limits, fracture toughness, seismic response, core support, simplified fatigue methods, inelastic piping design, thermal stratification design procedures
- **NRC Endorsement of Div 5 & associated Code Cases**
  - Strain Limits for Elevated Temp Service (Using E-PP Analysis)
  - Creep-Fatigue at Elevated Temp (Using E-PP Analysis)
  - Alloy 617

DOE Advanced Reactor Technologies R&D Supports Resolution of These Issues Plus Development & Qualification of Data Required for Design

Reviews for Advanced Reactors Found Shortcomings in High-Temp Metals & High-Temp Design Methodology (HTDM)

- **NRC/ACRS Review of Clinch River Breeder Reactor in mid-1980's [1]**
- **GE's PSID for PRISM 1986 – NRC Generated PSER in 1994 [2]**
- **ORNL Review for NRC of ASME Code Case N-47 (now NH and Div 5A) in 1992 [3]**
- **NRC Review and Assessment of Codes and Procedures for HTGR Components in 2003 [4]**
- **DOE-funded ASME/LLC Regulatory Safety Issues in Structural Design Criteria Review of ASME III NH in 2007 [5]**
- **NRC-sponsored Review of Regulatory Safety Issues for Elevated Temperature Structural Integrity for Next Generation Plants in 2008 [6]**

These reviews cumulatively identified over 40 individual concerns, but can be summarized under 8 key areas

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References for High Temperature Reactor Materials and Design Methods Reviews

1. Griffen, D.S., “Elevated-Temperature Structural Design Evaluation Issues in LMFBR Licensing,” Nuclear Engineering and Design, 90, (1985), pp. 299-306
2. NUREG-1368 “Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor,” Feb. 1994
3. NUREG/CR-5955, Huddleston, R.L. and Swindeman, R.W., “Materials and Design Bases Issues in ASME Code Case N-47,” ORNL/TM-12266, April 1993
4. NUREG/CR-6816, Shah, V.N., S. Majumdar, and K. Natesan,“Review and Assessment of Codes and Procedures for HTGR Components,” ATL-02-36, June 2003.
5. O’Donnell, W. J., and D. S. Griffin, “Regulatory Safety Issues in the Structural Design Criteria of ASME Section III Subsection NH and for Very High Temperatures for VHTR and Gen-IV,” ASME-LLC STP-NU-010, Dec. 2007
6. O’Donnell, W.J., Hull, A.B., and Malik, S., “Historical Context of Elevated Temperature Structural Integrity for Next Generation Plants: Regulatory Safety Issues in Structural Design Criteria of ASME Section III Subsection NH,” Proceedings of 2008 ASME Pressure Vessel and Piping Conf., PVP2008-61870, July 2008

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Ongoing High Priority ASME Code Committee Actions Endorsed by BNCS and Supported by DOE R&D Activities

Topics	2017 Edition	Beyond 2017
New simplified analysis methods (EPP) that replace current methods based on linear analysis (and can be used at higher temperatures)	X	
Adequacy of the definition of S values used for the design of Class B components, which is based on extrapolated properties at 100,000 hours, in light of application to 500,000 hours design	X	
Construction rules for “compact” heat exchanges		X
Incorporation of new materials such as Alloy 617 and Alloy 709 (austenitic stainless)	A617	A709
Pursuit of “all temperature code”		X
Complete the extension of Alloy 800H for 500,000 hr-design	X	
Complete the extension of SS304, 316 for 500,000 hr-design	X	
Complete the extension of Grade 91 for 500,000 hr-design	X	
Thermal striping		X
Develop design by analysis rules for Class B components (including compact HX)		X
Component classification (Refer back to ANS 53 classification rules), including assessment of: Is Class B really necessary?	X	
Add non-irradiated and irradiated graphite material properties		X

Chronology of Major Endorsement Efforts for ASME Section III Division 5

9/15 – Discussions at 1st DOE-NRC Workshop on Advanced Non-LWR Cooled Reactors on need for Div 5 endorsement

2/16 – Begin detailed discussions between NRO and DOE-NE on NRC plans for participation in relevant ASME Div 5 subgroups

3/16 – Endorsement by ASME BNCS of High Priority List for Div 5 Code Actions

6/16 – Presentation at 2nd DOE-NRC Workshop emphasizing need for and value of NRC endorsement of ASME Section III Division 5

2/17 – Two task Groups formed at ASME Code Week representing High Temperature Liquid- and Gas-Cooled Reactor working groups to define pathway and schedule for NRC endorsement of Div 5

- **Metallic structures & components**
- **Non-metallic support structures**

ASME Section III Division 5 Needs to Be Updated and Endorsed

- **There is a recoginized need to expand use of consensus-based codes and standards in the advanced reactor regulatory framework to minimize time to completion, provide flexibility in implementation, and enhance regulatory surety**
- **Discussions between NRC's Office of New Reactors, DOE-NE's Office of Advanced Reactor Technologies, and ASME's SubGroup on High Temperature Reactors have initiated the process for NRC to evaluate and eventually endorse Division 5**
- **A lack of NRC endorsement of ASME construction rules for advanced non-LWRs represents a significant regulatory risk that delays development & deployment and discourages commercial interest. It must be resolved.**

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Points of Contact for Additional ASME Div 5 Information

- **DOE Activities**
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  - **Metals and Design Methods - Sam Sham: ssham@anl.gov**
  - **Graphite – Will Windes: william.windes@inl.gov**
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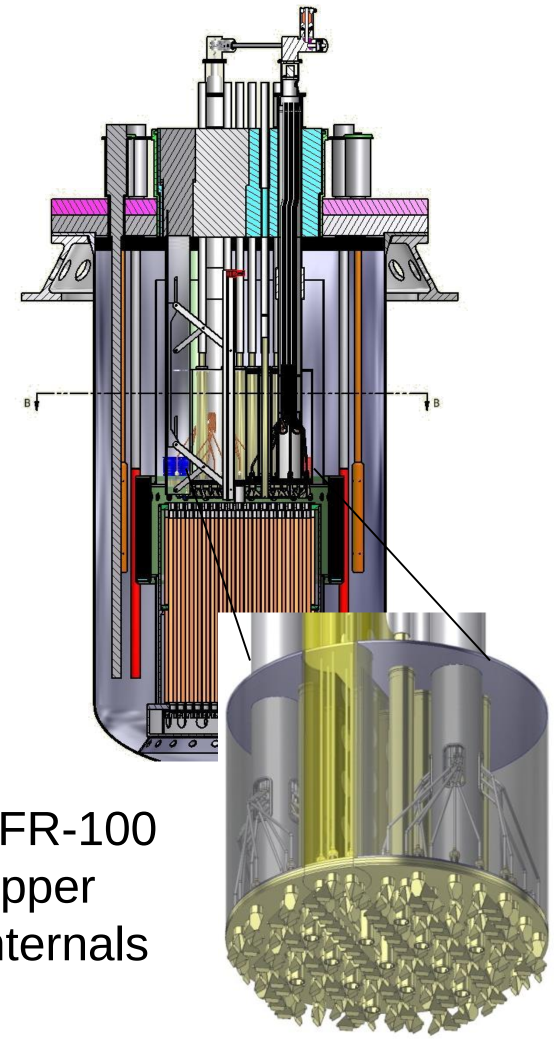
  - **Composites - Yutai Katoh: katohy@ornl.gov**

- **NRC Review for Endorsement**
  - **George Tartal: george.tartal@nrc.gov**
  - **Steven Downey: steven.downey@nrc.gov**
  - **Matthew Mitchell: matthew.mitchell@nrc.gov**

Examples of ASME Code Issues and Supporting DOE R&D

Examples of Active High-Temp Code Issues: More Accurate Predictions of Creep Fatigue Interaction Are Needed

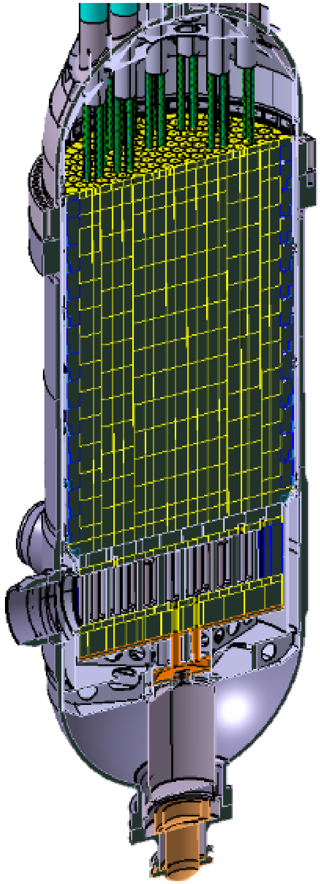
- **Understanding creep-fatigue interactions are critical for design and safety analysis of components in high temperature reactors**
- **Critical for components operating under inelastic conditions of time, temperature, and loading**
- **Cumulative damage from creep and fatigue must be understood and predicted**
- **Current ASME design rules are largely overconservative**



AFR-100 Upper Internals

Examples of Active High-Temp Code Issues: Negligible Creep Limits for RPV Materials Are Not Fully Defined

- **In LWR RPV design, no time-dependent deformation is considered, hence no creep-fatigue interactions are required**
- **For advanced reactors, RPVs will operate for longer times (60-year design) and possibly higher temperatures, hence negligible creep is a potential concern**
- **Negligible creep may impact cyclic performance and render design stress values non-conservative**



HTGR RPV

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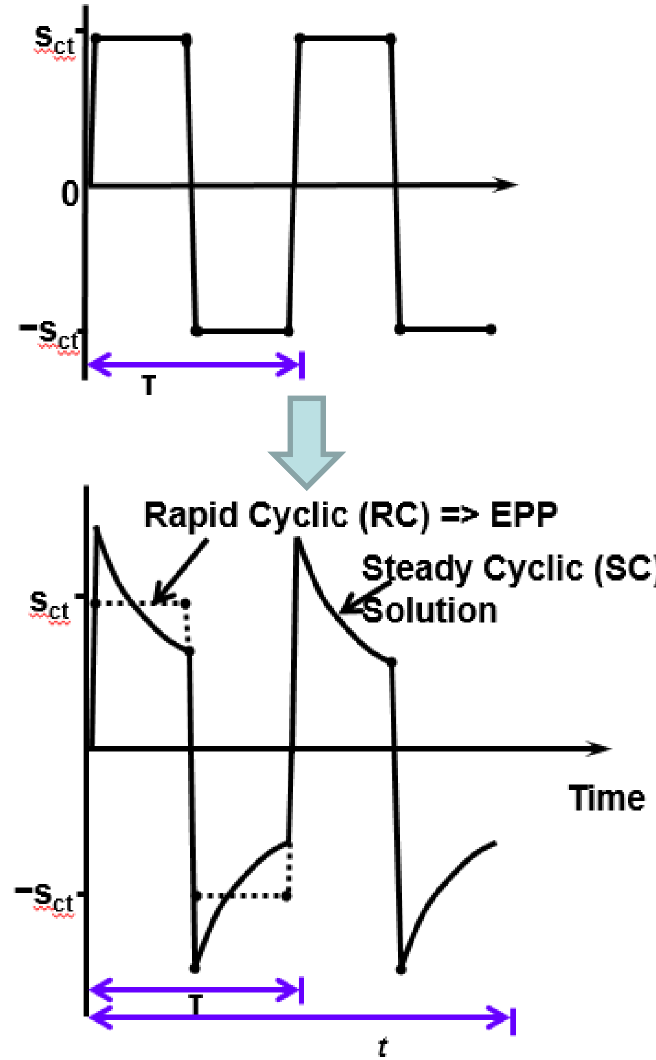
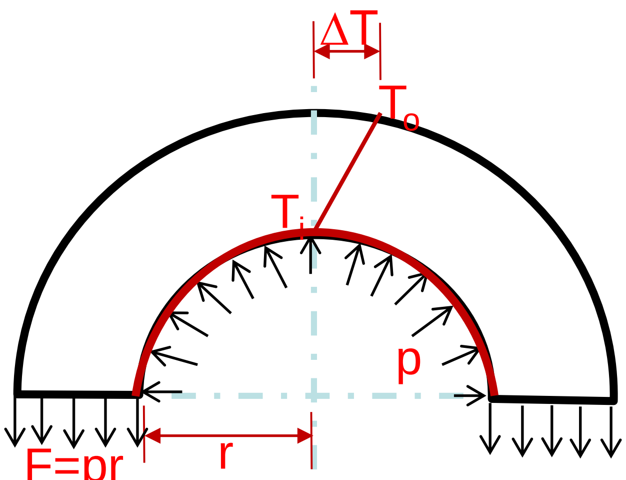
Improved Components of High-Temperature Design Methodology Being Developed

- **Code Cases for improved design rules, based on elastic-perfectly plastic analysis, approved for strain limits & creep-fatigue**
  - **Critical for very high temperatures where no distinction exists between creep and plasticity**
    - **Current rules invalid at very high temperatures**
    - **Will enable simplified methods for Alloy 617 > 1200°F (649°C)**
  - **E-PP analysis addresses ratchetting & shakedown**
  - **Avoids stress classification**
- **Yield strength is a "pseudo" strength given by the limiting design parameter, e.g. stress for 1% inelastic strain**
- **The Rapid Cycle (RC) is limiting case that bounds the real Steady Cyclic (SC) solution**

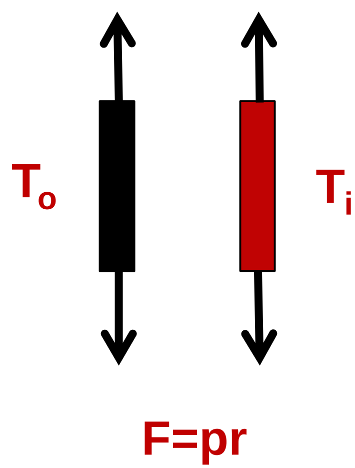


Advances in High Temperature Design Methodology Are Being Validated through Key Features Tests

- **Two-bar tests** can simulate combined thermal transients and sustained pressure loads that can generate a ratchet (progressive deformation) mechanism during creep-fatigue, relaxation, elastic follow-up, etc.
  - Validation of the E-PP model under varying effects of thermal path and mean stress



Pressurized cylinder with radial thermal gradient



Equivalent Two-bar model

- Equal deformations
- Pressure stress in vessel wall represented by total load on bars;
- Through-wall temperature gradient represented by temperature difference between bars

Additional High-Temperature Alloys, Now Being Qualified, Will Provide Additional Options for Nuclear Construction

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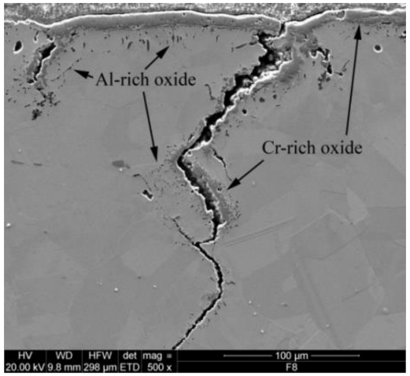
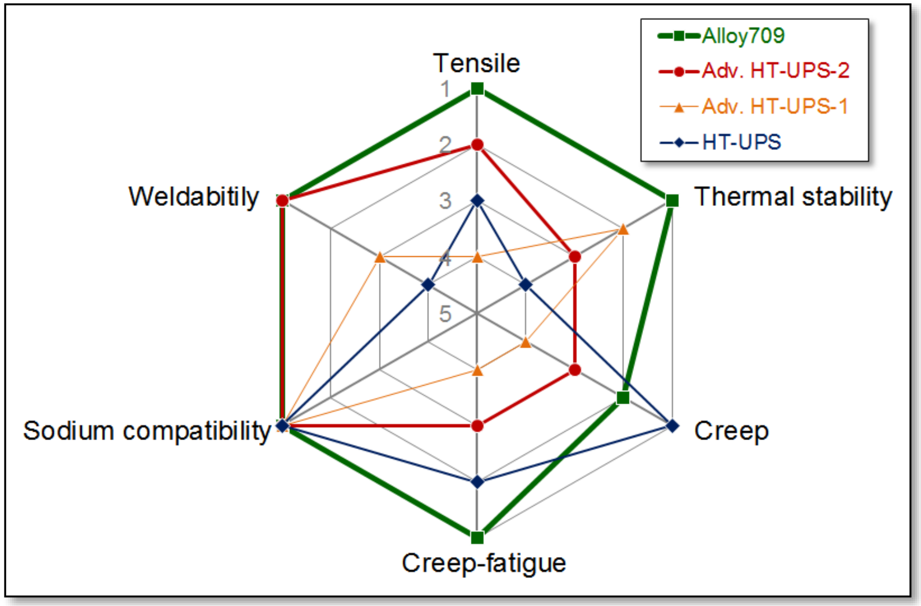
- **Alloy 617 Code Case being approved**
  - Advanced gas reactor heat exchangers & steam generators up to 950°C and 100,000 hrs
  - Low-temperature Code Case (T < 427°C) approved and high-temperature CC approval in progress
  - Anticipate inclusion in 2019 edition of Sec III Div 5



Creep-fatigue crack 617

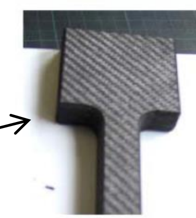
- **Alloy 709 selected for Code qualification**
  - Will provide improved performance, design envelop, and cost reduction for LMRs
  - Roughly double existing creep strength of existing stainless steels in Sec III Div 5
  - Qualification testing begun



Technical Bases for Code Rule Development of Graphite and Ceramic Composites Continuing to Expand

- **Graphite used for core supports in HTGRs, VHTRs and FHRs**
  - **Maintain core geometry and protect fuel**
  - **Includes current and future nuclear graphites**
- **Special graphite considerations for Code rules**
  - **Lack of ductility**
  - **Need for statistically set load limits**
  - **Requires irradiation and oxidation data**
- **Ceramic composites (e.g. SiC-SiC) for internals & controls for gas, liquid-metal & salt cooled systems**
  - **Very high temperature and irradiation resistance**
  - **Dose$_{max}$ > 100 dpa, T$_{max}$ ≥ 1200°C**
  - **Materials specification, design, properties, testing, examination, and reporting rules developing**

Graphite Core Supports



Core Tie Rod